Entergy

Entergy Operations, Inc.
P.O. Box B
Killona, LA 70066
Tel 504 739 6242

Early C. Ewing, III
Director
Nuclear Safety & Regulatory Affairs
Waterford 3

W3F1-98-0150
A4.05
PR

August 17, 1998

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Reporting of Licensee Event Report

Gentlemen:

Attached is Licensee Event Report (LER) 98-014 for Waterford Steam Electric Station Unit 3. This report provides details of a manual reactor trip, followed by the actuation of the Emergency Feedwater Actuation System that initiated Emergency Feedwater flow to both SGs. Emergency Feedwater flow control valve logic to SG#1 malfunctioned, requiring manual operator action to control SG level. This LER is submitted pursuant to 10CFR50.73(a)(2)(iv).

If you have any questions concerning this LER, please contact me at (504) 739-6242 or M.K. Brandon at (504) 739-6254.

Very truly yours,

E.C. Ewing
Director
Nuclear Safety & Regulatory Affairs

ECE/RLW/rtk

(w/Attachment)

cc: E.W. Merschoff (NRC Region IV)
C.P. Patel (NRC-NRR)
A.L. Garibaldi
J.T. Wheelock - INPO Records Center
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Administrator - LRPD

1165

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
WATERFORD STEAM ELECTRIC STATION UNIT 3	05000 382	1 OF 11

TITLE (4)
MANUAL REACTOR TRIP WITH SUBSEQUENT EFAS ACTUATION

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
07	16	98	98	-- 014 --	00	08	17	98	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 92

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
M.K. Brandon, Acting Licensing Manager	(504) 739-6254

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
Condens	JK	142X180RBG1	Gen Elec	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On July 16, 1998, at approximately 1100 CDT, while operating in Mode 1 at 92% reactor power, the reactor was manually tripped due to rapidly decreasing levels in both steam generators (SG) caused by a reduction in Feedwater Pump Turbine (FWPT) 'A' speed. Following the manual reactor trip, Emergency Feedwater Actuation System (EFAS) was actuated by SG low level, initiating Emergency Feedwater flow to both SGs. Emergency Feedwater (EFW) flow control valve logic to SG#1 malfunctioned, requiring manual operator action to control SG level. The root cause of the reduction in FWPT 'A' speed and loss of Steam Generator Feedpump (SGFP) was attributed to a short circuit in the FWPT 'A' speed control relay due to condensation. The most probable root cause of the EFW flow control valve logic malfunction was an intermittent failure of a mercury wetted nuclear relay card (NRC) in the EFW flow control valve level control mode circuit. The manual reactor trip and EFW flow control valve logic malfunction condition did not compromise the health and safety of the public.

REPORTABLE OCCURRENCE

On July 16, 1998 at approximately 1100 CDT, while operating in Mode 1 at 92% reactor power, the reactor was manually tripped due to rapidly decreasing levels in both steam generators (SG) [AB]. The decreasing SG level was caused by a reduction in Feedwater Pump Turbine FWPT 'A' speed [JK-TRB] and the loss of Steam Generator Feedpump (SGFP) [SK-P]. Following the manual reactor trip, Emergency Feedwater Actuation System (EFAS) [JE] was actuated by SG low level (27.4% narrow range with steam generator pressure greater than 764 psia), initiating Emergency Feedwater [BA] flow to both SGs. Emergency Feedwater (EFW) flow control valve logic to SG#1 malfunctioned, requiring manual operator action to control SG level. These conditions are reportable in accordance with 10CFR50.73(a)(2)(iv) as an event or condition that resulted in a manual actuation of the Reactor Protection System (RPS) and automatic actuation of an Engineered Safety Feature (ESF).

INITIAL CONDITIONS

On July 16, 1998, at approximately 1000 CDT, a reactor downpower from 100% to 92% was commenced to perform repairs on main turbine governor valve #4 [SB-V]. At the time of the event, Waterford 3 was stabilized at 92% reactor power. There was no major equipment out of service or Technical Specification LCOs in effect specific to this event.

EVENT DESCRIPTION

On July 16, 1998, at approximately 1000 CDT, while operating in Mode 1 at 100% reactor power, the following sequence of events occurred (all times approximated):

1000 Operations personnel commenced plant downpower to 92% rated thermal power (RTP) to repair electro-hydraulic system [TG] leaks on main turbine governor valve #4.

1025 Operations personnel completed plant downpower to 92% RTP.

1055 Operations received annunciators for SGFP A Flow Low, SGFP A Flow Low-Low, SG 1&2 Stm Flow/Feed Flow Signal Deviation, and SG 1 and 2 Level High/Low. Levels in both SGs started rapidly decreasing. FWPT 'A' speed decreased to approximately 3200 RPM from a normal operating speed of 5000 RPM at 100% RTP. FWPT 'A' governor steam valves [SK-V] failed closed due to loss of the speed control signal. SG level continued a rapid decrease to 30% narrow range.

1102 Operations personnel initiated a manual reactor trip. Immediately following the reactor trip, automatic EFAS-1 and EFAS-2 actuation signals (Emergency Feedwater Actuation Signal) [JE] were received (27.4% narrow range with steam generator pressure greater than 764 psia). SG levels continued to decrease to below 55% wide range. EFW injection commenced, supplying feedwater to both SGs. Over the next 12 minutes, EFW flow control valve logic functioned properly and continued recovering SG levels to approximately 68% wide range.

1115 With SG#1 level recovering to 68% wide range, EFW flow to SG#1 increased from approximately 450 gpm to greater than 800 gpm. The controller demand output signal for backup EFW flow control valve EFW223A [BA-FCV] was 60%, whereas, the primary EFW224A EFW flow control valve [BA-FCV] controller demand output signal was only 20%. The nuclear plant operator took manual control of both EFW223A and EFW224A EFW flow control valves and closed EFW223A. Operations declared SG#1 EFAS control valve logic inoperable and entered TS 3.3.2, Table 3.3-3, functional unit 7.e.

1118 Both SG levels were restored to normal levels, between 50-70% narrow range, with SG#2 EFW flow control valve controllers in automatic and SG#1 EFW flow control valve controllers in manual control.

1126 EFW flow control valves were closed and both SGs were being supplied by SGFP 'B'.

Loss of FWPT 'A' Speed Control Investigation

Following the manual reactor trip, Plant Management established a Significant Event Response Team (SERT) to investigate the loss of FWPT 'A' speed control. The initial investigation into the loss of FWPT 'A' speed control found the following visual evidence: (1) 'A' Steam Generator Feedwater Pump Turbine speed control cabinet [JK] indicator lights were not illuminated, indicating a loss of power; (2) a distinctive odor associated with failed electronic components and visible damage was found on the speed control relay card; (3) presence of water droplets on the failed relay card and a visible water stain proceeding from the top edge of the card directly to the damaged section on the card; and (4) the presence of two small water droplets at a conduit outlet located directly above the card, however no evidence of water was found in the bottom of the cabinet.

A more detailed inspection was performed to identify the source of the water droplets. An inspection of the cabinets and conduits located in the Exciter Switchgear Room found the presence of moisture in several of the conduits leading to the Feedpump Control cabinet. The inspection was performed initially and repeated following an afternoon rain. The rain did not cause additional water to appear. The conduit routing would not likely provide a passage of rainwater into the cabinet. The source of water droplets was apparently due to condensation of water vapor in the conduits routed from an area of high ambient temperature (100°F) and high relative humidity into the Exciter Switchgear Room Feedpump Control cabinet. The conduits are open to the Turbine Building [NM] and the Feedpump Control cabinet. The Exciter Switchgear Room was air conditioned and normally maintained at 69°F. The environmental conditions of the various areas further support the conclusion that the source of the water droplets was from condensation in the conduit.

EFW Control Valve Logic Malfunction Investigation

Plant Management established a Significant Event Response Team (SERT) to investigate the EFW control valve logic malfunction of SG#1. The team established the areas of investigation and verified that: (1) EFAS signals (27.4% narrow range with steam generator pressure greater than 764 psia) to both SGs were valid ESF actuations; and (2) both SG levels dropped below 55% wide range and EFW injection to the SGs were valid.

The third area of investigation involved the EFW control valve logic response to SG level changes during the EFW actuation. The system and control valve logic response during an EFW actuation is designed such that the normal EFW system injection flow to each steam generator is through two parallel paths, each containing one FCV and one isolation valve. The EFW system injection flow rates into the steam generators are controlled based on SG wide range levels. As steam generator level decreases to 55% wide range (critical level) the primary flow control valve opens to a preset position equivalent to 200 gpm total flow. The 200 gpm flow rate is based upon SG pressure at 1000 psia and two EFW motor driven pumps running. If the primary valve does not supply sufficient flow, the backup FCV will open as required to provide 175 gpm total flow. The total flow to each SG is between 175 and 200 gpm. If the SG level were to decrease to 45% wide range, the primary FCV maintains the preset position equivalent to 200 gpm. The backup FCV will open as required to provide a total flow of 400 gpm to each steam generator. At 36.3% wide range, a priority open

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

signal is generated to fully open the primary and backup FCV to supply maximum flow to each SG.

The priority open signal is reset by 40% wide range level and the primary FCV returns to its preset position (approximately 200 gpm). The backup FCV modulates to supply 400 gpm total. When the 68% level is achieved, the primary FCV will maintain its preset position (200 gpm) and the backup FCV transfers to the Level Control Mode. As the level increases past the 68% wide range level, the backup FCV modulates until fully closed. As steam generator level increases to 71% wide range, the primary FCV transfers to the Level Control Mode. Level is maintained at 71% wide range level by the modulating primary FCV.

The investigation revealed that SG#2 control valve logic response to the SG level changes functioned as designed. SG#2 level was restored to the normal 50-70% narrow range level. However, SG#1 EFW control valve logic response did not function as designed. During SG#1 level recovery, the backup flow control valve (FCV) EFW223A should have automatically transferred from the Flow Control Mode to the Level Control Mode when 68% wide range level was achieved. As the SG level increased past the 68% wide range level, the backup FCV should have modulated closed. At 68% wide range level, EFW flow to SG#1 increased from approximately 450 gpm to greater than 800 gpm. The controller demand output signal for backup EFW flow control valve EFW223A was 60%, when it should have been considerably less and decreasing to 0%. The primary EFW224A EFW flow control valve was at the expected controller demand output signal of 20%. The plant nuclear operator took manual control of EFW223A and the flow control valve responded as designed to manual operation.

The Team evaluated the potential failure modes of backup EFW flow control valve EFW223A that would cause the valve to unexpectedly open at a point where the control logic circuit would have the valve modulate to the fully closed position. The potential failure modes consisting of valve failure, loss of instrument air, and manual open signal to the valve were evaluated and eliminated as sources of the malfunction. The

EFW223A control circuit was analyzed and troubleshot. It was determined that the most probable failure mode was the failure of a mercury wetted relay on the Nuclear Relay Card [BA] in the control circuit. All electronic components of the control circuit were functionally tested and revealed no problems. However, the Nuclear Relay Card in the control circuit contained mercury-wetted relays [BA-RLY] that were unique to the level control mode. The mercury-wetted relays transfer the control signal from flow control to level control mode. If the level control relay did not close and the signal was absent, flow control valve EFW223A would fully open. The relay card was replaced and functionally tested satisfactorily in the EFW223A control loop prior to plant startup.

CAUSAL FACTORS

Loss of FWPT 'A' Speed Control

The root cause of the reduction in FWPT 'A' speed and loss of 'A' SGFP was attributed to condensation falling on the FWPT 'A' speed control relay card which caused a short circuit and subsequent loss of control power to the FWPT 'A' Speed Control System. The failure mode was an external damaging source to the component that resulted in failure of the relay card.

The loss of the FWPT 'A' speed control signal caused the FWPT 'A' governor steam valves to fail closed which resulted in the decrease in SGFP speed and loss of 'A' SGFP. Feedwater flow was reduced which caused the SG levels to decrease rapidly. The Feedwater Control Systems [JB] responded appropriately to the decreasing SG levels and increased demand for feedwater flow which called for the Startup Feedwater Regulating Valves [SJ-V] and the Main Feedwater Regulating Valves [SJ-V] to full open and both FWPT speed demand signals to increase. The associated Feedwater Control valves and Feedpump Turbines responded as expected, given the failure of the FWPT 'A' relay card in the speed control circuit.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
WATERFORD STEAM ELECTRIC STATION UNIT 3	05000 382	98	-- 014	-- 00	8 OF 11

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

EFW Control Valve Logic Malfunction

The most probable root cause of the EFW223A flow control valve logic malfunction was an intermittent failure of a mercury-wetted relay on the NRC in the logic control circuit. Industry operating experience revealed that intermittent failures of this type have occurred such that the relay can fail to change state, yet once cycled, the relay will operate properly.

The failure of this mercury wetted level control relay to close caused flow control valve EFW223A to fully open. The failure of EFW223A in the open direction resulted in an increase in flow through EFW223A to SG#1 from a logic controlled flow of 450 gpm to over 800 gpm. However, manual control of the FCV EFW223A remained available to the nuclear plant operator.

Plant conditions during the event may have contributed to the magnitude of the flow changes. As discussed earlier, the primary FCV maintains a preset position of 200 gpm, that is based upon steam generator pressure at 1000 psia and two EFW motor driven pumps running. During the event, the plant had all three EFW pumps running, steam generator pressure less than 1000 psia, and the estimated flow rate through the primary FCV was greater than 400 gpm. These actual plant conditions resulted in much greater EFW flow available with a higher differential pressure between the EFW system and the steam generator.

CORRECTIVE MEASURES

Loss of FWPT 'A' Speed Control

Immediate Actions Following the Trip and Prior to Plant Startup:

1. Plant Management established a Significant Event Response Team (SERT) to determine the cause of the FWPT speed control problem.

2. Completed troubleshooting the FWPT governor control circuitry, which included the supporting relay cards and power supplies, and replaced all damaged components resulting from the relay card short circuit.

3. The conduits entering the FWPT Control cabinet were sealed to prevent future condensation intrusion.

4. The failed 'A' FWPT speed control relay card was replaced.

5. An inspection of 'B' FWPT speed control cards and circuit connections was completed to ensure no abnormalities existed.

6. Conduits entering other air-conditioned enclosures that contain Trip Sensitive equipment were inspected to ensure they were not vulnerable to similar failure modes.

Long Term Actions Following Plant Startup

1. I&C Maintenance shall perform a failure analysis of the speed control relay card to identify failed components on the card.

2. I&C Maintenance shall perform periodic follow-up inspections within the Feedpump Control cabinet to verify sealed conduits have eliminated the condensation intrusion.

EFW Control Valve Logic Malfunction

Immediate Actions Following the Trip and Prior to Plant Startup:

1. Plant Management established a SERT to determine the cause of the EFW control valve logic malfunction.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

2. Completed troubleshooting the EFW EFW223A control valve logic, which included the supporting relay cards and power supplies, and reviewed the past surveillance data.

3. Replaced the NRC card, considered the most probable cause of the EFW flow control valve logic malfunction.

4. Performed a functional test to verify proper function of EFW223A.

Long Term Actions Following Plant Startup

1. System Engineering shall perform additional industry history searches through Operational Engineering Experience Group to determine if there have been experiences similar to this event.

SAFETY SIGNIFICANCE

Manual Reactor Protection System Actuation

The manual reactor trip was initiated due to rapidly decreasing levels in SGs caused by a reduction in MFWPT 'A' speed. The event is bounded by the Loss of Normal Feedwater Flow event that is analyzed in UFSAR Section 15.2.2.5. The plant response to the loss of SGFP 'A' with the resulting decrease in feedwater flow was normal. The EFAS actuated a few seconds after the manual reactor trip occurred to maintain the steam generators as heat sinks. This event did not represent any actual safety significance.

EFW Control Valve Logic Malfunction

The safety function of the Emergency Feedwater system is to provide cooling water to one or both steam generators for the purpose of decay heat removal from the reactor coolant system (RCS) in response to any loss of main feedwater to the steam generators. The system is designed to perform its intended function in the event of a single failure in the system. The system functioned as designed for approximately 12 minutes during the level recovery of SG#1, until a malfunction of the flow control valve logic occurred in one of two redundant flow paths to SG#1. This single failure generated an excessive EFW flow rate that could have potentially caused an overfeeding of SG#1. Prompt operator manual action mitigated the potential consequences of overfeeding the SG and prevented potential moisture carryover from the steam generators into the steam lines. The operator took manual control of the malfunctioning flow control valve and closed the valve. The remaining EFW flow path to SG#1 remained operable and capable of providing sufficient cooling water to remove RCS decay heat. This event did not represent any actual safety significance.

In conclusion, the plant and subsequent operator action responded to the manual reactor trip due to rapidly decreasing levels in both steam generators as required. The appropriate EFAS actuation signal was generated and EFW actuation occurred when required. Although EFW flow to SG#1 was larger than expected, prompt operator action precluded overfeeding SG#1. The plant was controlled and maintained throughout this event such that the health and safety of the general public and plant personnel was not compromised. Therefore, these events did not represent any actual safety significance.

SIMILAR EVENTS

A review of Waterford 3 LERs dating back to June 1996 was performed. There were no reports found which revealed any generic or recurring problems with FWPT speed control relay card or EFW control valve logic NRC card failures.

ADDITIONAL INFORMATION

The EIIS component function identifier is included in brackets [ ] following the first reference to a system component.

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9808190191 DOC.DATE: 98/08/17 NOTARIZED: NO DOCKET #
FACIL:50-382 Waterford Generating Station, Unit 3, Louisiana Power 05000382
AUTH.NAME AUTHOR AFFILIATION
EWING,E.C. Entergy Operations, Inc.
RECIP.NAME RECIPIENT AFFILIATION
Records Management Branch (Document Control Desk)

SUBJECT: Forwards LER 98-014-00 for Waterford Steam Electric Station, Unit 3.Rept provides details of manual reactor trip,followed by actuation of EFW actuation sys,initiated EFW flow to both SGs.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 2+11
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD4-1 PD	1	1	PATEL,C	1	1
INTERNAL:	ACRS	1	1	AEOD/SPD/RAB	2	2
	AEOD/SPD/RRAB	1	1	FILE CENTER	1	1
	NRR/DE/ECGB	1	1	NRR/DE/EELB	1	1
	NRR/DE/EMEB	1	1	NRR/DRCH/HICB	1	1
	NRR/DRCH/HOHB	1	1	NRR/DRCH/HQMB	1	1
	NRR/DRPM/PECB	1	1	NRR/DSSA/SPLB	1	1
	RES/DET/EIB	1	1	RGN4 FILE 01	1	1
EXTERNAL:	L ST LOBBY WARD	1	1	~~LITCO BRYCE,J H~~	1	1
	NOAC POORE,W.	1	1	NOAC QUEENER,DS	1	1
	NRC PDR	1	1	NUDOCS FULL TXT	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 23 ENCL 23